AD-A094 067 NAVAL RESEARCH LAB WASHINGTON DC F/G 18/10

THE NRL-EPRI RESEARCH PROGRAM (RP886-2), EVALUATION AND PREDICT--ETC(U)

DEC 80 J R HAWTHORNE

UNCLASSIFIED NRL-MR-4431-PT-1 NL

1 OF 1

AD A094067

END

DATE FILMED

2-81

DTIC

The NRL-EPRI Research Program ([illegible]), Evaluation and Prediction of Neutron Embrittlement in Reactor Pressure Vessel Materials, Annual Progress Report for CY 1979

Part I. Dynamic C_V, PCC_V Investigations

J.R. Hawthorne, Editor

Thermostructural Materials Branch
Material Science and Technology Division

December 31, 1980

NAVAL RESEARCH LABORATORY
Washington, D.C.

SECURITY CLASSIFICATION OF THIS PAGE (When Data Entered)

REPORT DOCUMENTATION PAGE

READ INSTRUCTIONS BEFORE COMPLETING FORM

1. REPORT NUMBER: NRL Memorandum Report 4431	2. GOVT ACCESSION NO.: AD-A094 067
3. RECIPIENT'S CATALOG NUMBER	
4. TITLE (and Subtitle): THE NRL-EPRI RESEARCH PROGRAM (RP886-2), EVALUATION AND PREDICTION OF NEUTRON EMBRITTLEMENT IN REACTOR PRESSURE VESSEL MATERIALS, ~~ANNUAL PROGRESS REPORT FOR CY 1979~~, PART I. DYNAMIC ~~C_V, PCC, INVESTIGATIONS~~	5. TYPE OF REPORT & PERIOD COVERED: Part I, Final Report
	6. PERFORMING ORG. REPORT NUMBER
7. AUTHOR(s): J. R. Hawthorne, Editor	8. CONTRACT OR GRANT NUMBER(s)
9. PERFORMING ORGANIZATION NAME AND ADDRESS: Naval Research Laboratory, Washington, DC 20375	10. PROGRAM ELEMENT, PROJECT, TASK AREA & WORK UNIT NUMBERS: 63-1069-0-1; RP886-2
11. CONTROLLING OFFICE NAME AND ADDRESS: Electric Power Research Institute, Palo Alto, CA 94304	12. REPORT DATE: December 31, 1980
	13. NUMBER OF PAGES: 34
14. MONITORING AGENCY NAME & ADDRESS (if different from Controlling Office)	15. SECURITY CLASS. (of this report): UNCLASSIFIED
	15a. DECLASSIFICATION/DOWNGRADING SCHEDULE

16. DISTRIBUTION STATEMENT (of this Report)

Approved for public release; distribution unlimited.

17. DISTRIBUTION STATEMENT (of the abstract entered in Block 20, if different from Report)

18. SUPPLEMENTARY NOTES

Prepared for the Electric Power Research Institute (EPRI), under Agreement RP886-2.

19. KEY WORDS (Continue on reverse side if necessary and identify by block number)

Charpy-V test	Nuclear reactors	Steel weldments
J integral	Precracked Charpy test	
R curve	Pressure vessels	
Fracture toughness	Radiation embrittlement	
Low alloy steels	Steel forgings	
Notch ductility	Steel plate	

20. ABSTRACT (Continue on reverse side if necessary and identify by block number)

Nuclear reactor pressure vessel materials are subject to progressive reductions in fracture resistance in service due to neutron irradiation. Current technology is inadequate to quantitatively predict radiation embrittlement for all vessel materials and their metallurgical variations for the neutron fluences of interest. In addition a relationship between apparent notch ductility and fracture toughness in the irradiated condition is needed to evolve more quantitative projections of structural integrity. The NRL-EPRI RP886-2 Program was formulated to advance both areas for the benefit of reactor vessel design and operation. A primary objective of this program is the development of a high-quality data base for the evaluation of current radiation-embrittlement projection methods and the development of improved methods.

(Continues)

DD FORM 1473, 1 JAN 73 EDITION OF 1 NOV 65 IS OBSOLETE
S/N 0102-LF-014-6601

SECURITY CLASSIFICATION OF THIS PAGE (When Data Entered)

20. Abstract (Continued)

This report documents program highlights and research results for CY 1979 along with plans for the completion of program investigations. Postirradiation test data are presented for plate, forging ~~and~~ weld deposit materials irradiated in six reactor experiments to fluences ranging from ~0.1 to ~3x10^{19} n/cm^2 >1 MeV at 288°C. Comparisons are made between results for standard Charpy V-notch and fatigue precracked Charpy-V tests of preirradiation and postirradiation material conditions. A companion document (Annual Progress Report for CY 1979: Part II) ~~to be published~~ will present results for the 25.4 mm compact toughness (J-R curve) tests of the same materials and material conditions. A preliminary correlation of the Charpy-V and J-integral fracture toughness property changes with irradiation is observed.

CONTENTS

THE NRL-EPRI RESEARCH PROGRAM (RP886-2),
EVALUATION AND PREDICTION OF NEUTRON EMBRITTLEMENT
IN REACTOR PRESSURE VESSEL MATERIALS,
ANNUAL PROGRESS REPORT FOR CY 1979
Part I. Dynamic C_v, PCC_v Investigations

SUMMARY

The NRL-EPRI RP886-2 Research Program was formalized in January 1977. This report documents the highlights and accomplishments of the NRL investigations during CY 1979, the third year of program operation.

The program focuses on radiation-induced changes to reactor pressure vessel materials typical of past as well as current commercial production. Radiation effects are being investigated by standard Charpy-V (C_v), fatigue precracked Charpy-V (PCC_v), and compact toughness (CT) test methods. Primary accomplishments during CY 1979 relate to the completion of material irradiation experiments, the continuation of postirradiation material evaluations, and the evolution of mechanical property correlations.

Highlights of investigations made during this reporting period include:

- Development of the first experimental comparison of notch ductility and fracture toughness at a low (48-J) C_v postirradiation upper shelf level.

- Development of data supporting earlier tentative indications of a general agreement between the radiation-induced elevation in C_v 41-J and K_{Jd} 99MPa m transition temperatures. The suggested correlation, if confirmed by remaining post-irradiation assessments, will be of major value to the interpretation and use of C_v results from reactor vessel surveillance.

- Observation of large differences between measured postirradiation C_v 41-J transition temperature elevations and projections of irradiation-induced transition temperature change for high copper content weld materials computed from NRC Regulatory Guide 1.99. Comparisons among program materials indicate that the Guide is not equally conservative for all materials and fluence conditions evaluated.

Additional research highlights pertaining to CT test investigations are described in Part II of this Annual Progress Report.

INTRODUCTION - J. R. Hawthorne

The cooperative research and development program between the Electric Power Research Institute (EPRI) and the Naval Research Laboratory (NRL) is generally directed at materials and material applications for nuclear energy systems, with research emphasis on material reliability and environmental responses for system safety. The present effort focuses on the degradation of fracture resistance of reactor vessel steels and weld metals by a 288°C (550°F) radiation environment.

Manuscript submitted November 17, 1980.

Properties under study are notch ductility, fracture toughness, and strength. Objectives are (a) to develop a data base for the evaluation of current radiation-embrittlement projection methods and for the development of improved procedures, (b) to investigate the relationship, if one exists, between radiation effects measured by the C_v test method and fracture mechanics test methods, (c) to determine the radiation embrittlement sensitivities of a broad range of reactor pressure vessel materials (plates, forgings, welds), and (d) to experimentally assess the effects of selected composition variations.

This annual report summarizes achievements of the third year of program effort. Previous reports [1,2] described program progress in CY 1977 and CY 1978, respectively. Program investigations are scheduled for completion in CY 1980.

MATERIALS - J. R. Hawthorne

Materials and Test Matrix

Program materials are identified in Table 1. The bases for materials selection were discussed in Ref. 1. In general, test specimens were taken from the quarter thickness location in the wrought materials and from through-thickness locations in the weld deposits. The transverse (TL, weak) test orientation was selected for study for the plate and forgings. Weld metal specimens were oriented to make the plane of fracture perpendicular to the weldment surface and parallel to the welding direction.

Table 2 illustrates the materials irradiation test matrix. Each of the experiments included standard C_v specimens for notch ductility determinations and PCC_v and 25.4 mm thick CT specimens for fracture toughness determinations. The three fluence levels represent, respectively, initial vessel service, early life and end of life conditions. During this reporting period, the material previously listed as "undesignated" for experiment 12 (BSR-14) was selected. That is, a decision was made to evaluate the longitudinal (LT, strong) orientation of the A302-B plate, Code N, in view of its TL orientation performance with the aim of comparing J-integral fracture toughness properties of one material over a wide range of C_v upper shelf levels (47 to 107-J).

Table 1 - Program Materials

Material	Code	Type
A533-B plate	CAB	U. S. production
A533-B plate	CBB	Foreign production
A302-B plate	N	U. S. production (ASTM A302-B reference)
A508-2 forging	BCB	U. S. production
S/A* weld 1	E19	High copper content (~0.35% Cu) and low upper shelf energy (USE)
S/A weld 2	E24	High copper content and high USE
S/A weld 3	E23	Intermediate copper content (~0.20% Cu) and low USE
S/A weld 4	E4	Intermediate copper content and high USE

*Submerged arc weld process.

Table 2 - Irradiation Test Matrix

Material*	Irradiation Assessments		
	1-2 x 10^{18} n/cm^2 > 1 MeV	≈ 8 x 10^{18} n/cm^2 > 1 MeV	3-4 x 10^{19} n/cm^2 > 1MeV
A533-B plate (CAB)		• (BSR-2)*	• (BSR-3,5)
A533-B plate (CBB)			• (BSR-4)
A302-B plate (N)			
TL Orientation			• (BSR-7)
LT Orientation			• (BSR-14)+
A508-2 forging (BCB)			• (BSR-6)
Weld 1 (E19)	• (BSR-9)	• (BSR-8)	• (BSR-10)
Weld 2 (E24)		• (BSR-11,15)	
Weld 3 (E23)		• (BSR12)	
Weld 4 (E4)			• (BSR-13)

* Bulk Shielding Reactor, experiment 2.
\+ CY 1979 decision.

Specimen Machining

The C_v and PCC_v specimens were machined to the standard dimensions given by ASTM Recommended Practice E23 (type A specimens). The CT specimens were machined to the dimensions given in reference (2). Specifications for fatigue precracking of PCC_v and CT specimens provided an a/W ratio of 0.5. The maximum allowable stress intensity (K_I) during the last increment of fatigue crack growth, 0.76 mm (0.030 in.), was 22 MPa$\sqrt{m}$ (20 ksi$\sqrt{in.}$) and 24 MPa$\sqrt{m}$ (22 ksi$\sqrt{in.}$) for the PCC_v and CT specimens respectively. Specimen fatigue precracking records on file document fatigue cycles, stress intensity, R ratio, and surface crack lengths for all stages of precracking.

REACTOR FACILITY AND OPERATIONS - H. E. Watson and J. R. Hawthorne

Status of Material Irradiation Experiments

During this period, reactor exposures were completed for experiments BSR 6 through BSR 9, BSR 11, BSR 12 and BSR 15. Experiments under irradiation at the time of this report were BSR 10 and BSR 13. The irradiation of experiment BSR 14 (see preceding section) will commence in CY 80 to conclude all scheduled reactor operations (See Table 2).

Neutron Spectra Calculation

Computations of the neutron flux spectra for the EPRI-NRL irradiation facilities and experiment designs were completed. The spectrum calculations were performed by the Oak Ridge National Laboratory (ORNL) under contract to NRL. Results are documented in reference [3], and apply to the initial and the current fuel core configurations (e.g. cores 37 and 54 respectively) used for the EPRI-NRL experiments (See Figures 1 and 2). The core 54 computation (position 78) assumed the simultaneous

BSR CORE LOADING No. 54

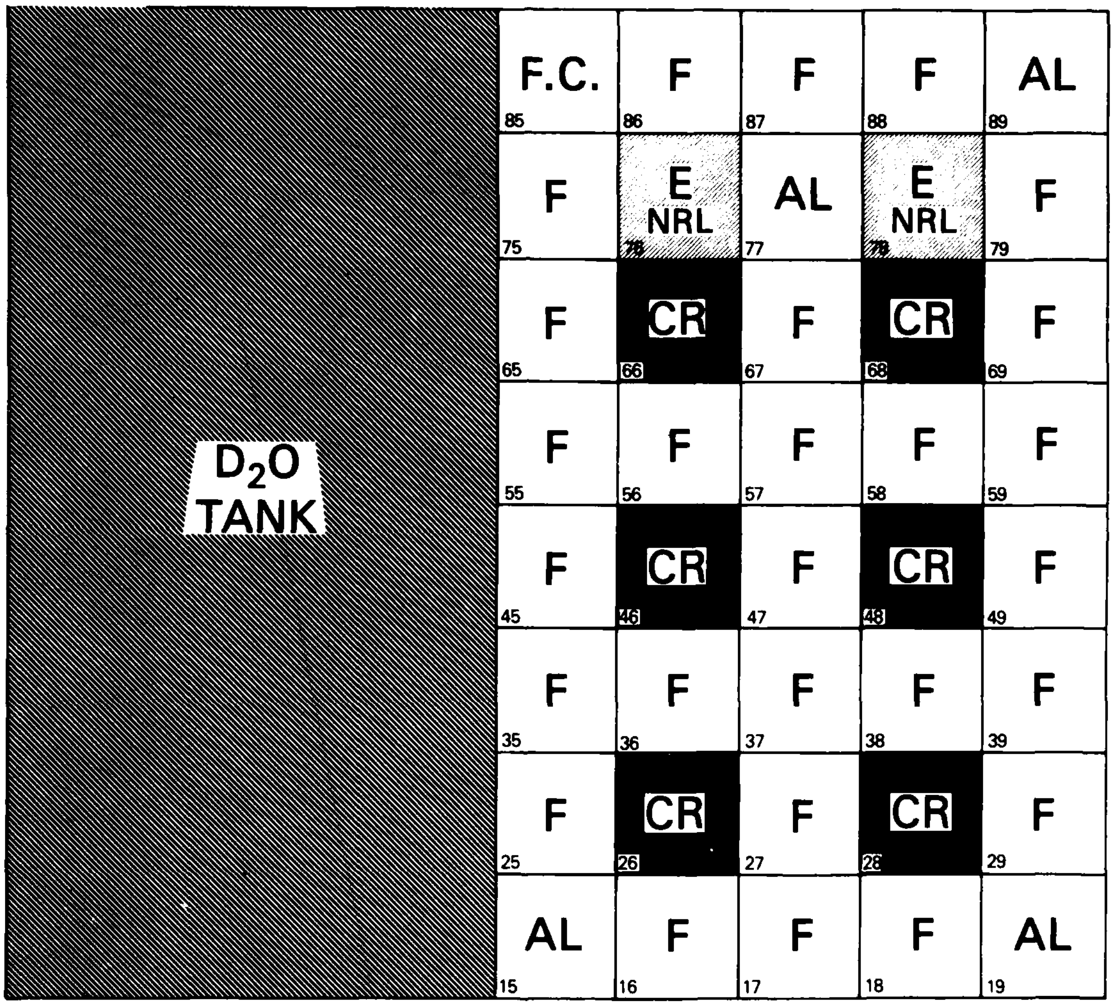

Fig. 1. Fuel core configuration in the BSR reactor for simultaneous NRL-EPRI experiment operations using core positions 76 and 78 [2]. The desiginations F, CR, FC and AL stand respectively for fuel element, control rod, fission chamber and aluminum piece.

BSR CORE LOADING No. 37

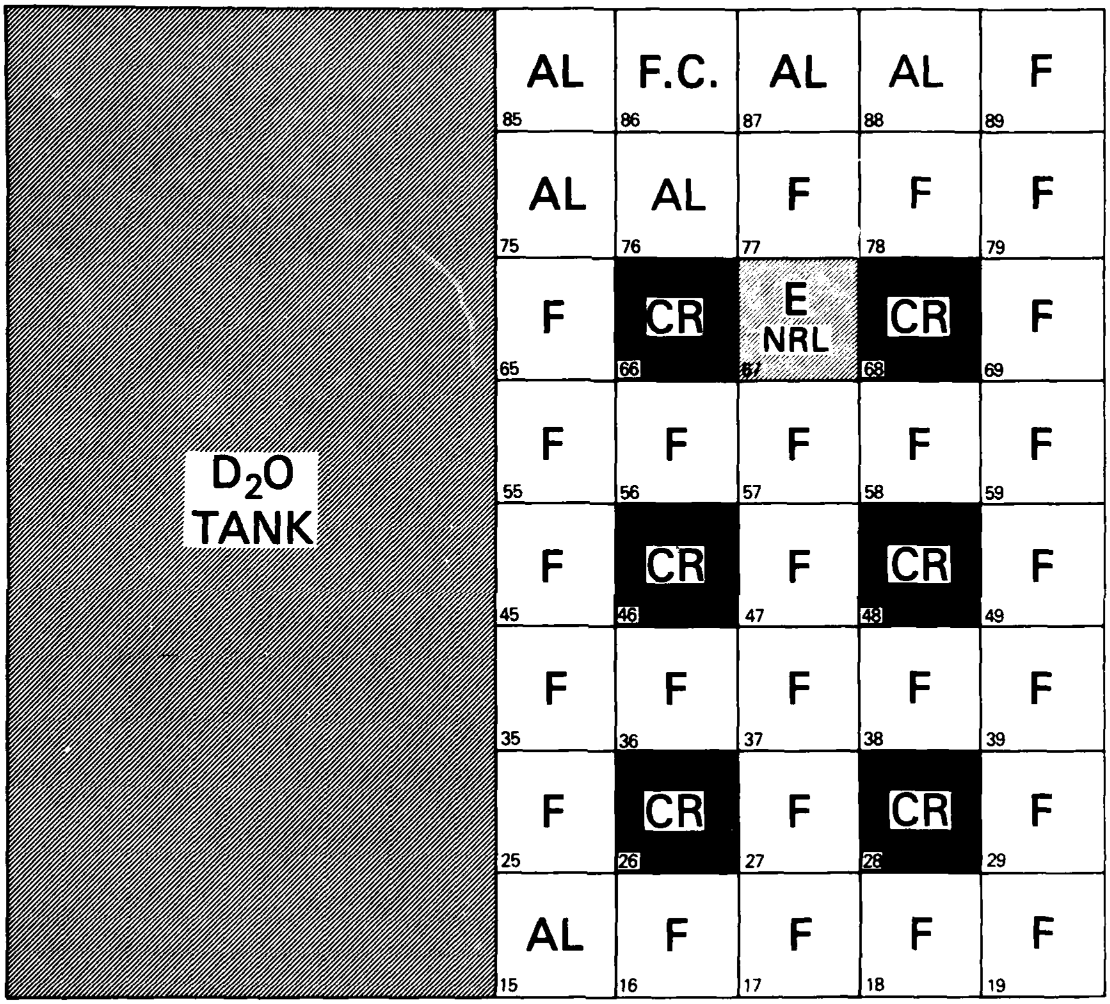

Fig. 2. Original fuel core configuration in the BSR reactor for NRL-EPRI experiment operations using core position 67 only.

use of two core positions (78 and 76) and the current (revised) experiment design. The core 37 computation (position 67) assumed the use of the original experiment design [2].

To summarize the ORNL findings, the calculated spectrum and assumed fission spectrum fluences for E>1MeV for the current core configuration are in the relation: $\Phi^{cs} = 0.954\,\Phi^{fs}$. Additionally, the calculated spectrum fluence for E>0.1MeV is approximately equal to $2.18\,\Phi^{cs}$ E>1MeV. In comparison, the calculations for core 37 gave the relations: $\Phi^{cs} = 1.08\,\Phi^{fs}$ (E>1MeV) and Φ^{cs} (E>0.1MeV) = $2.11\,\Phi^{cs}$ (E>1MeV).

During this period, 5 sets of dosimeters (20 wires each) were also irradiated in core position 78 using the NRL flux monitor assembly, for a further assessment of local BSR spectrum conditions by the Argonne National Laboratory (ANL) (R. Heinrich). Each dosimeter set consisted of gadolinium-covered and uncovered dosimeter wires. Reactor power was purposely reduced to 1 MW for the 72-hour long exposure. Results of this special neutron flux survey and analysis are not yet available.

FRACTURE TOUGHNESS TESTING - J. R. Hawthorne

Overview

The emphasis of this program is on the definition of toughness trends relative to the brittle-to-ductile transition regime. The research program permits limited studies of the upper shelf, however, as specimen numbers permit. To evolve a full analysis of data significance, irradiation data and trend information developed here will be evaluated statistically by a related EPRI program.

The primary focus of evaluations with the CT specimens is on fracture initiation toughness. However, it was expected that the materials would exhibit elastic-plastic behavior over the major portion of the brittle-to-ductile transition regime. Accordingly it was felt necessary to characterize the slow-stable crack extension phenomenon, commonly denoted by the R curve, as well. The latter is useful not only for defining crack initiation but also for assessing material potential for crack instability.

Charpy-V (C_v) and Precracked Charpy-V (PCC_v) Specimen Evaluations

Unirradiated Condition

In the previous report [2], preirradiation condition C_v data developed by NRL for the two A533-B plates, codes CAB and CBB, and the A508-2 forging, code BCB, were presented. Comparisons of these data against data generated by the RP232 Program [4] for other sections of the three materials revealed significant differences in two cases. Initially, RP232 Program C_v and PCC_v data were to serve for the reference condition for plates CAB, CBB, and N and the forging BCB. However, the NRL results confirmed the merit of conducting separate (check) tests as a precautionary measure. In the case of the newly fabricated weldments, codes E19, E23, and E24 and the production weldment, code E4, NRL is generating the primary reference condition data for its preirradiation versus postirradiation comparisons. (Welds E19, E23 and E24 are also included in an EPRI sponsored research program at Westinghouse.)

During this reporting period, preirradiation condition C_v data were developed for the four welds and for the particular section of A302-B plate employed. Additionally,

preirradiation condition PCC_v data were developed by NRL for the A302-B plate, the two A533-B plates, the A508-2 forging and the welds E23, E24 and E4. Reference PCC_v testing of weld E19, originally assigned elsewhere, has been given to NRL for completion in CY 1980.

NRL findings with preirradiation C_v and PCC_v tests are illustrated in the individual data figures and are summarized in tables 3 and 4. Referring first to the C_v data, several observations can be made as follows:

- A302-B plate, code N (Fig. 3). The results for the transverse test orientation show good agreement between quarter and midthickness test locations in the transition region and between the present data set and prior NRL test results [5]. A small difference (7-J) in upper shelf level between quarter and midthickness regions is observed; however, the upper shelf data for the one quarter and three quarter thickness layers match well.

- High copper, low upper shelf weld, code E19 (Fig. 4). Good agreement is observed between the data sets developed independently by NRL and Westinghouse. With the exception of the temperature interval between -1°C and 4°C (30 and 40°F), the data scatter is relatively small. The data spread at ~4°C could be the result of a weld layer effect. That is, the two "high" energy values represent layer 4 and layer 5 specimens (one quarter thickness region) while the data points located on the "average" trend line between -18°C and 27°C (0°F and 80°F) represent specimens from layers 12 through 16. (Note: Specimens were taken in 16 layers through the weld thickness).

- High copper, high upper shelf weld, code E24 (Fig. 5). Data sets developed by NRL and Westinghouse show significant differences in the upper half of the brittle/ductile transition region. However, the data sets are in quite good agreement in their definition of 41-J and 68-J transition temperatures and upper shelf energy level. Additional tests may be undertaken if clarification is required.

- Intermediate copper, low upper shelf weld, code E23 (Fig. 6). Good agreement between data sets is noted. In parallel with the data trend for weld E19, the upper shelf level of this weld rises with increasing temperature. This characteristic has been observed for several other A533-B submerged arc welds made with Linde 80 welding flux [6].

- Intermediate copper, high upper shelf weld, code E4 (Fig. 7.) A consistent difference in upper shelf level, but not transition behavior, is described by the data for the one quarter versus three quarter thickness regions of this production weld. A similar difference in fracture toughness was not evident in the limited upper shelf PCC_v data developed thus for this weld (see Fig. 17 below).

Results of PCC_v tests of the reference condition are illustrated in Figures 8-11 and 13-17. Main observations are as follows:

- A533-B plates, codes CAB and CBB (Figs.8,9). Comparisons of NRL data against data from the RP232 Program indicate small but discernable differences between the respective test sections. The observed differences do not change the conclusions made earlier for these plates, and relative PCC_v vs C_v property changes with irradiation [2]. In the case of plate CAB, the PCC_v tests of NRL

Text continues on page 23

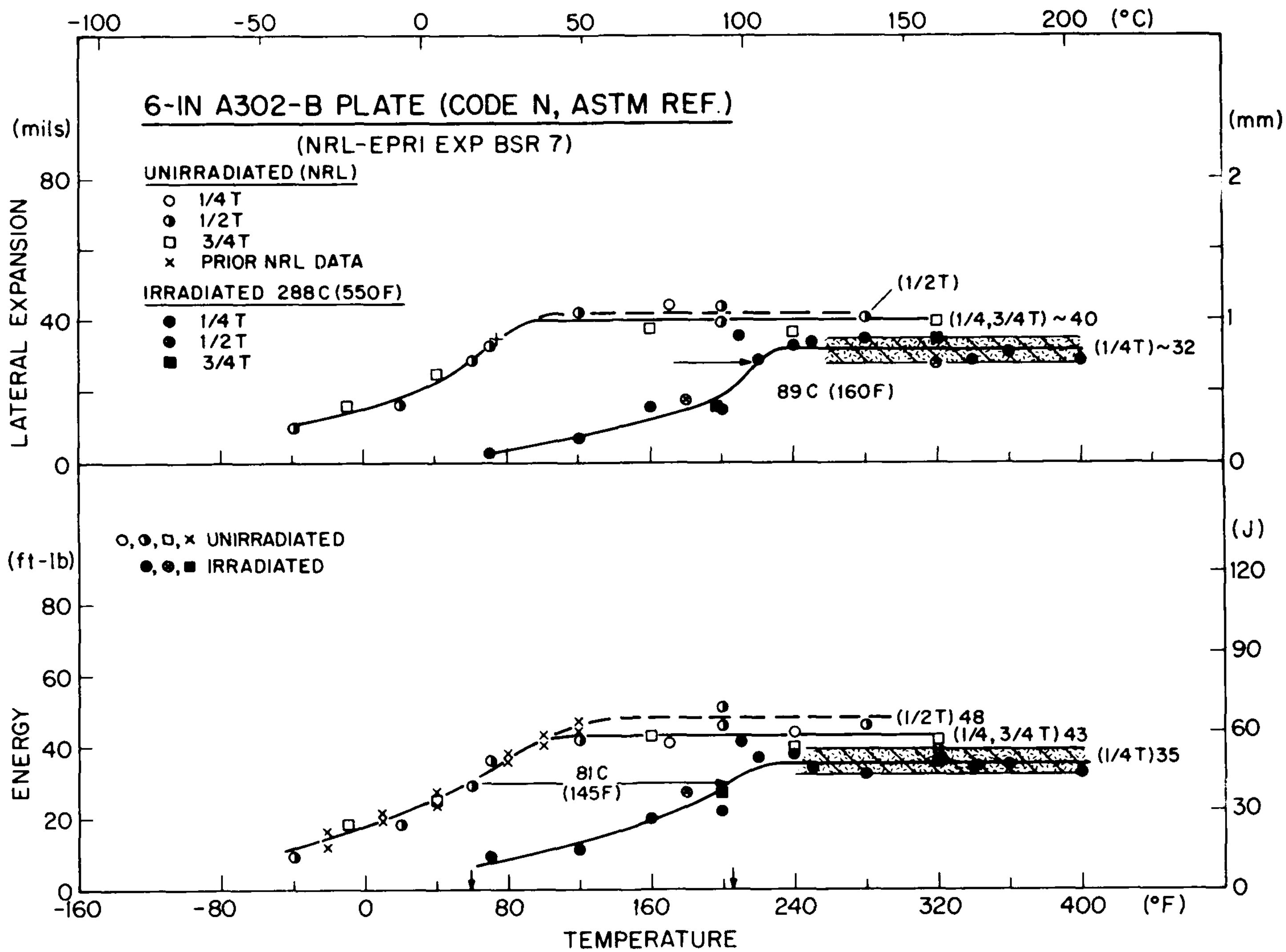

Fig. 3. Charpy-V notch ductility of A302-B plate, code N, before and after irradiation to an estimated fluence of $\sim 3 \times 10^{19}$ n/cm^2, E > 1 MeV (experiment BSR-7).

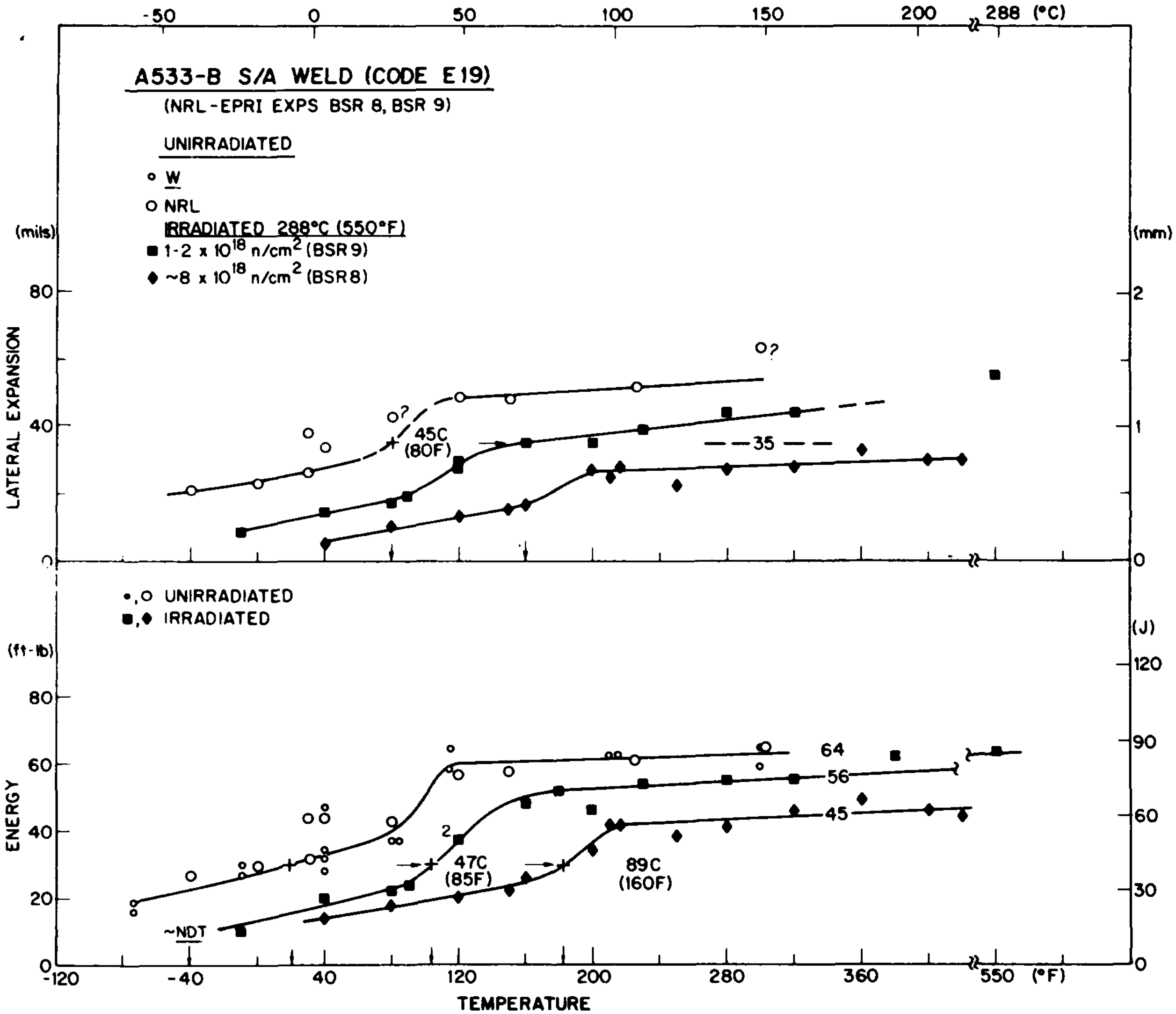

Fig. 4. Charpy-V notch ductility of submerged arc weld, code E19, before and after irradiation to two fluence levels (experiments BSR-8 and BSR-9).

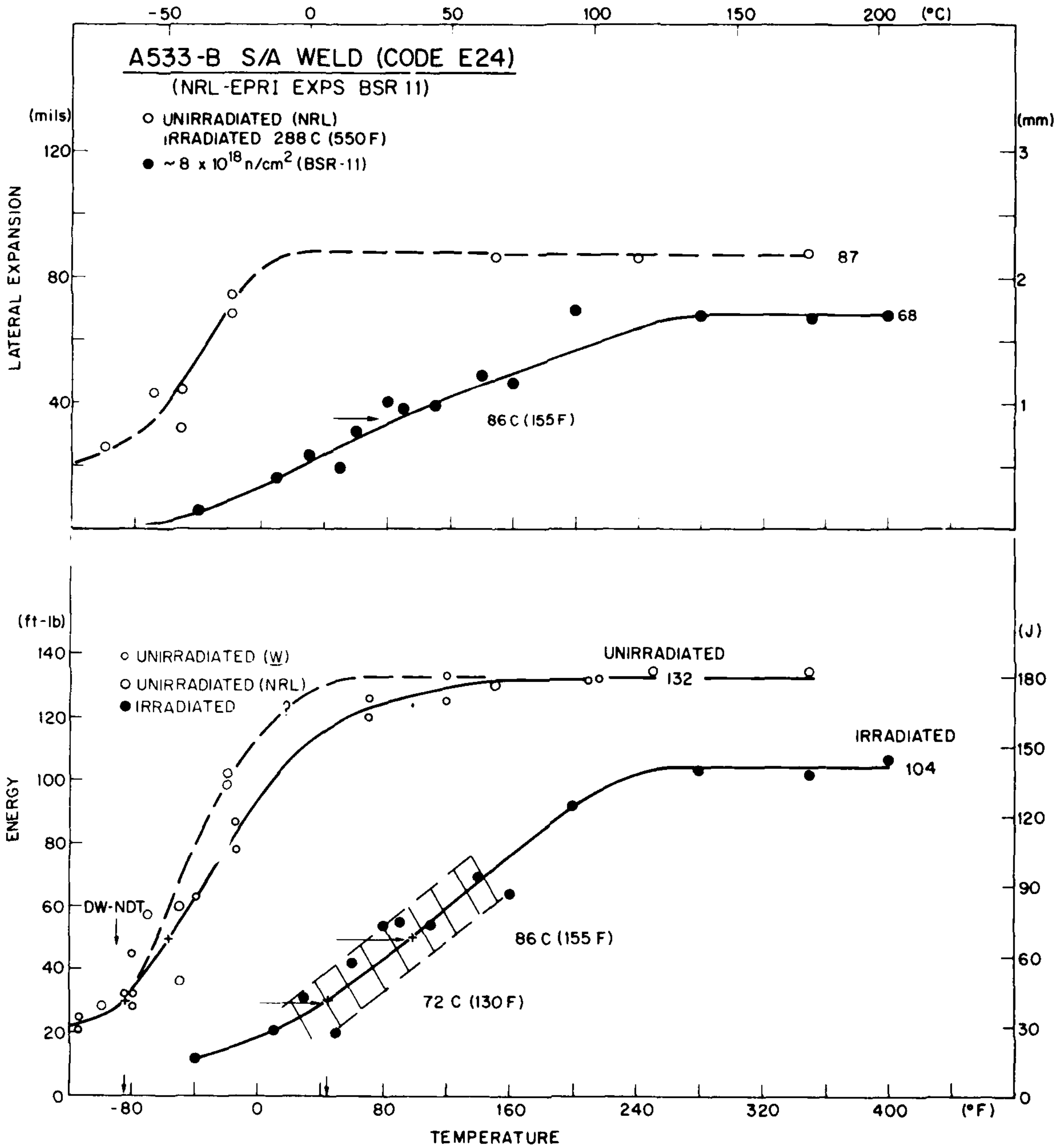

Fig. 5. Charpy-V notch ductility of submerged arc weld, code E24, before and after irradiation (experiment BSR-11). Additional tests of the preirradiation condition are planned.

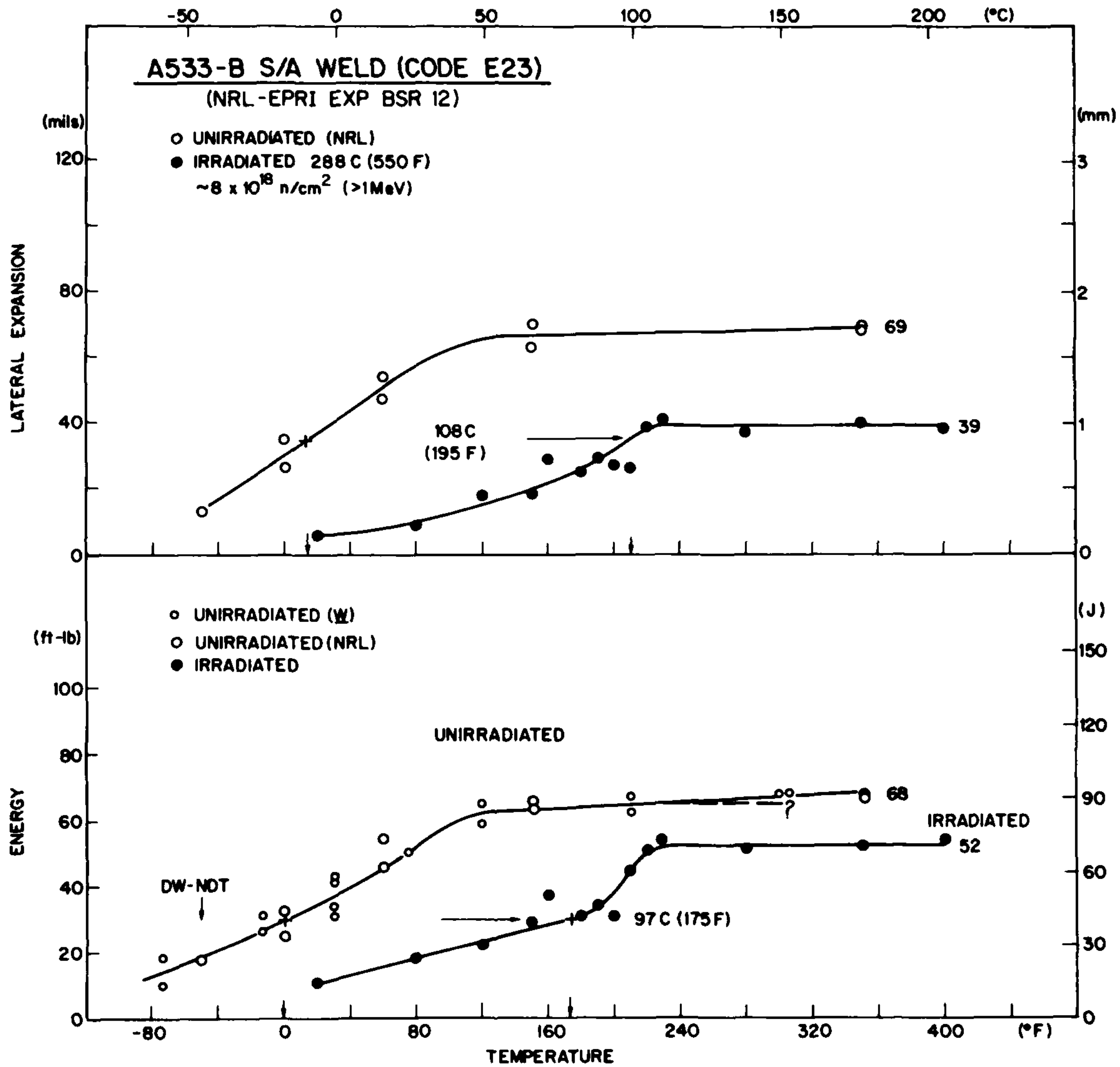

Fig. 6. Charpy-V notch ductility of submerged arc weld, code E23, before and after irradiation (experiment BSR-12).

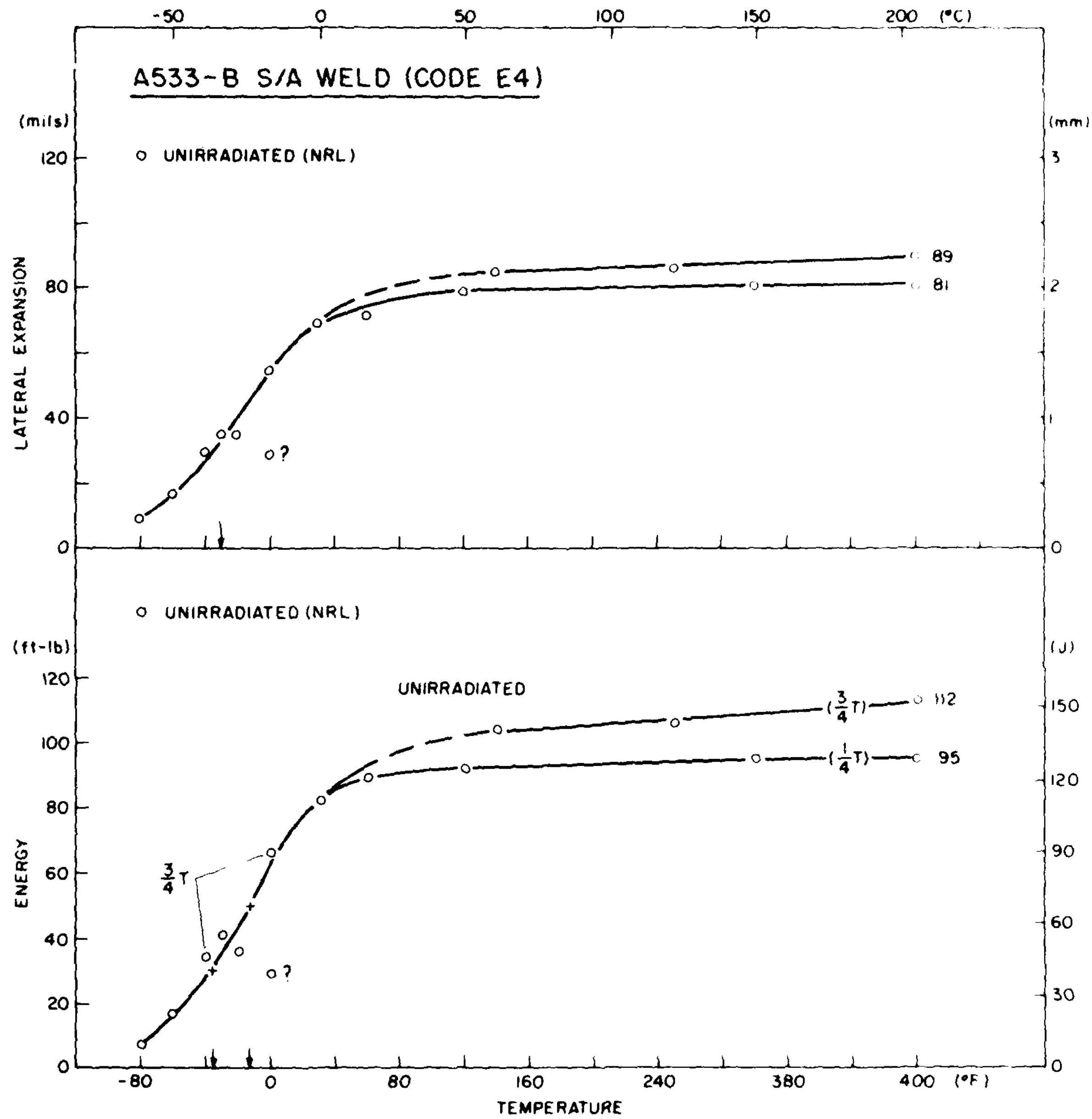

Fig. 7. Charpy-V notch ductility of submerged arc weld, code E4, before irradiation. Note the difference in upper shelf level between one quarter and three quarter thickness regions of this commerical production weld.

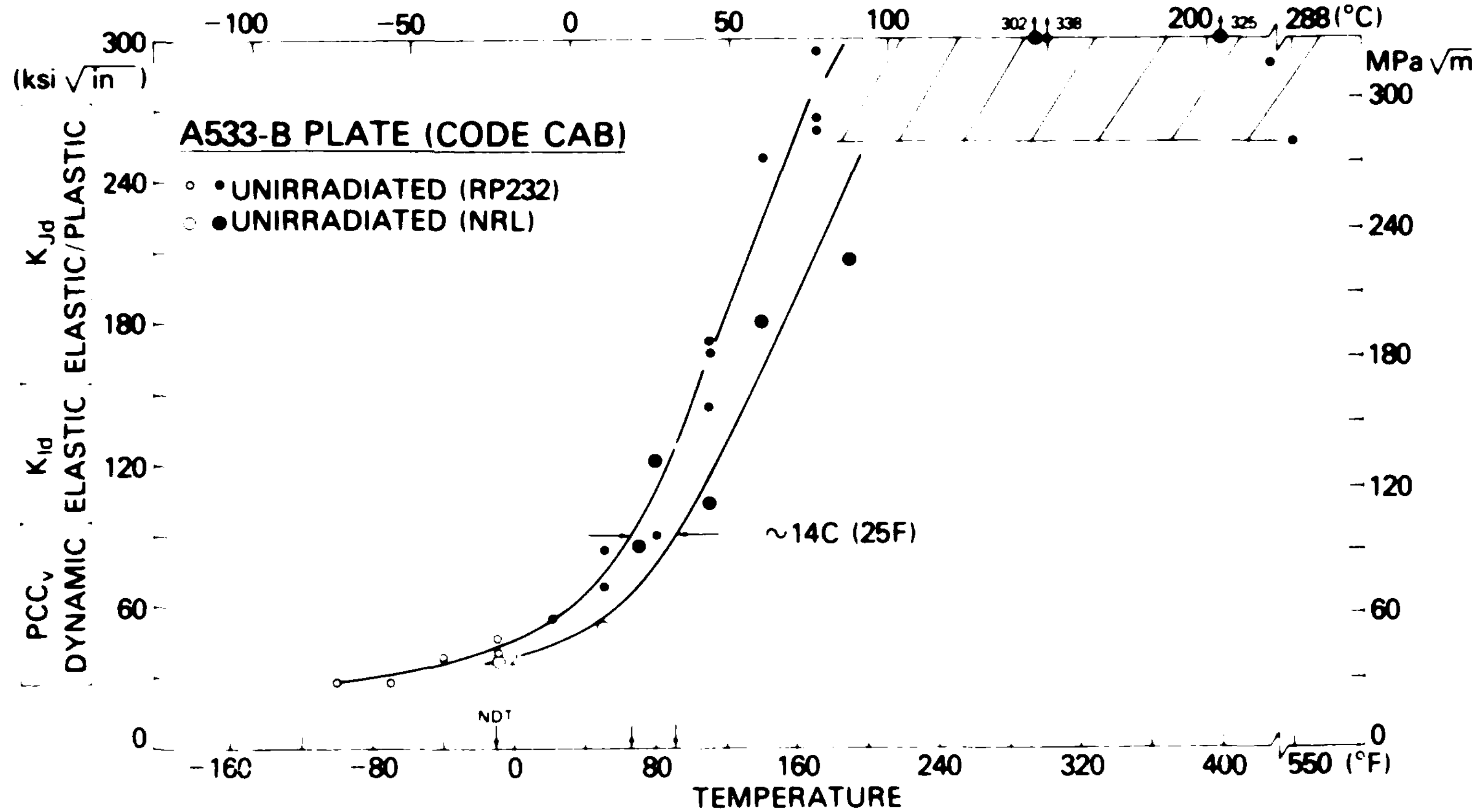

Fig. 8. Fatigue precracked Charpy-V test results for the A533-B plate, code CAB, before irradiation.

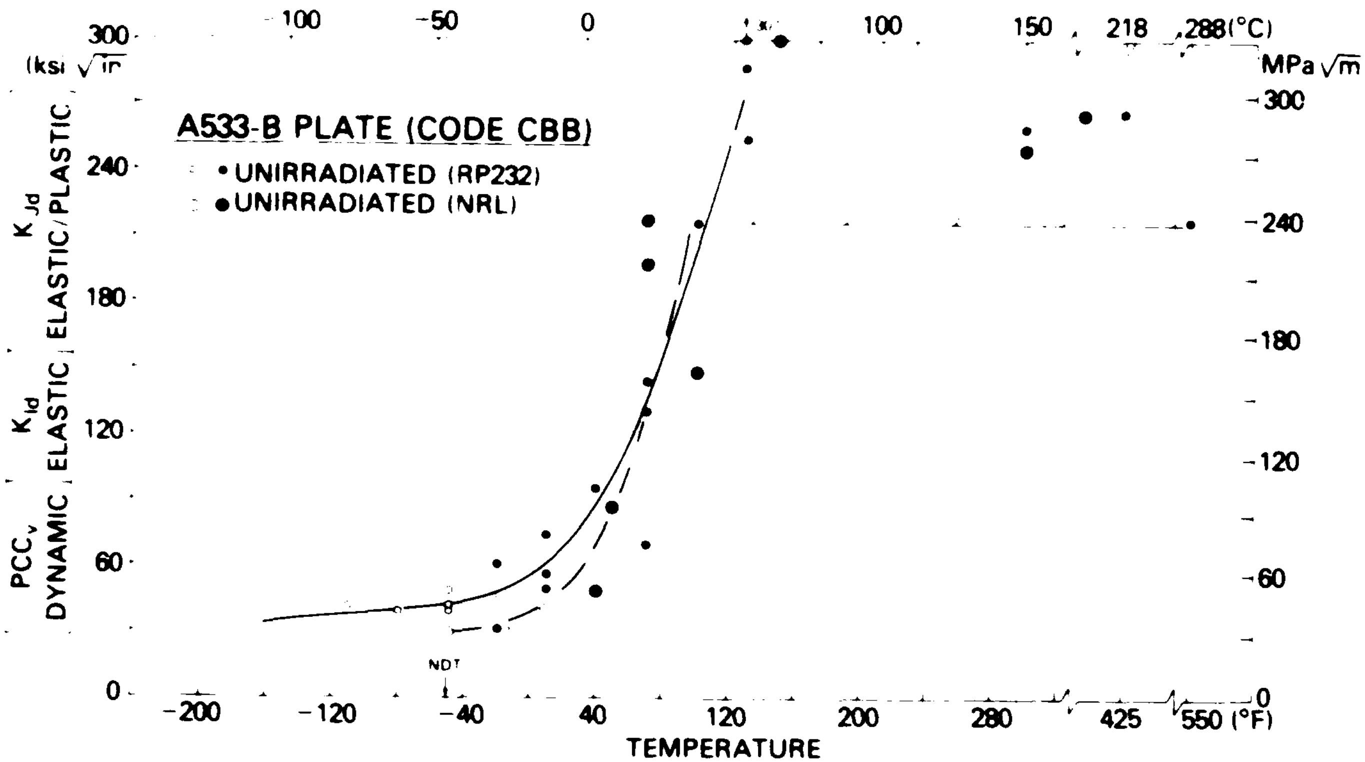

Fig. 9. Fatigue precracked Charpy-V test results for the A533-B plate, code CBB, before irradiation.

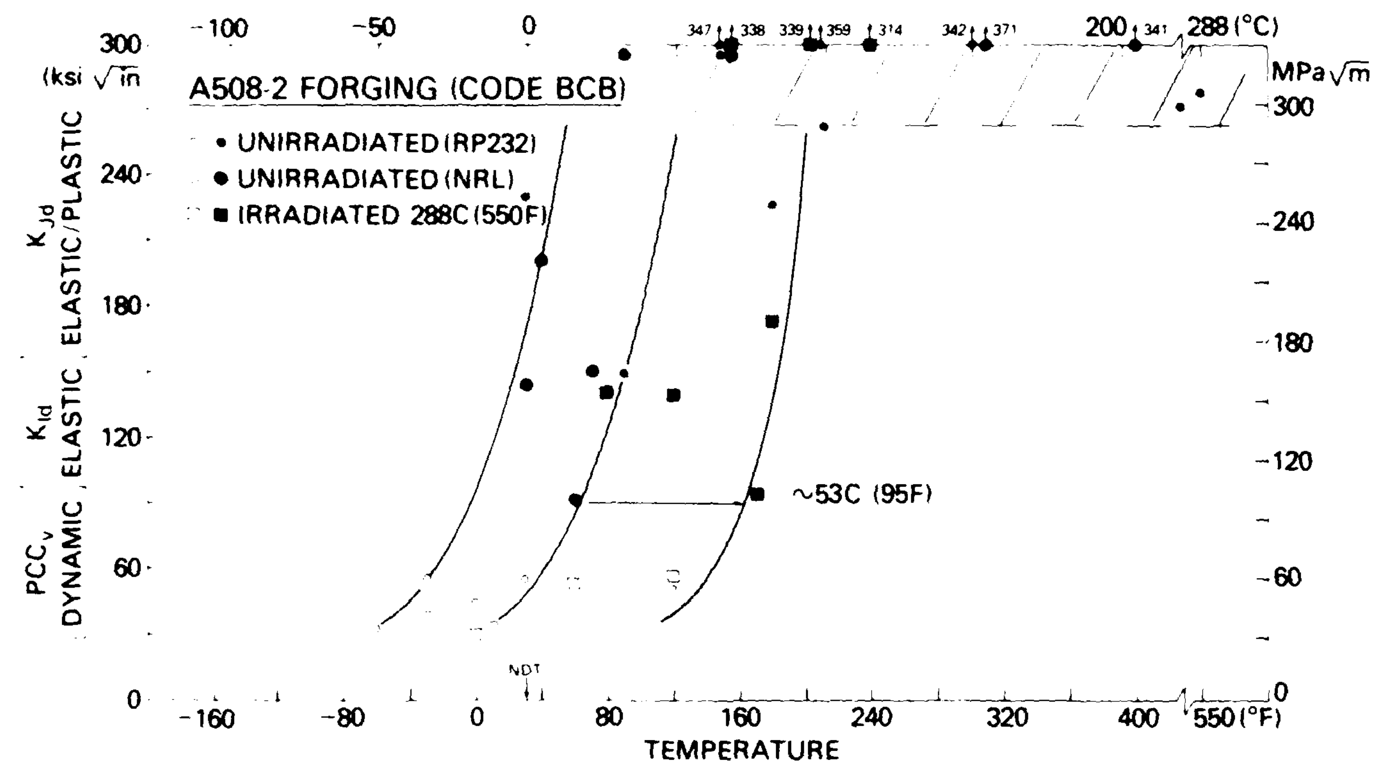

Fig. 10. Fatigue precracked Charpy-V test results for the A508-2 forging, code BCB, before and after irradiation. (experiment BSR-6). Note the wide scatter in data for the individual test conditions.

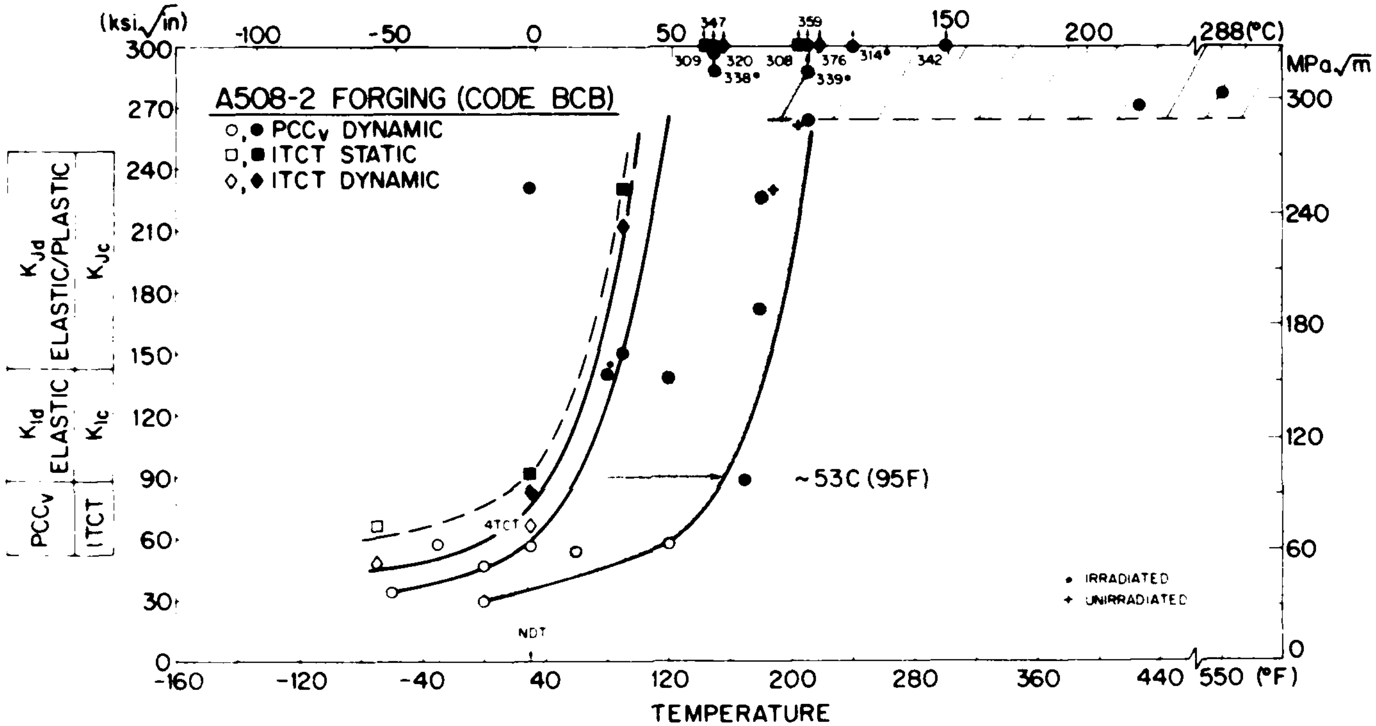

Fig. 11. Fatigue precracked Charpy-V test results for the A508-2 forging, code BCB, after irradiation referenced to the RP232 Program test results only (experiment BSR-6).

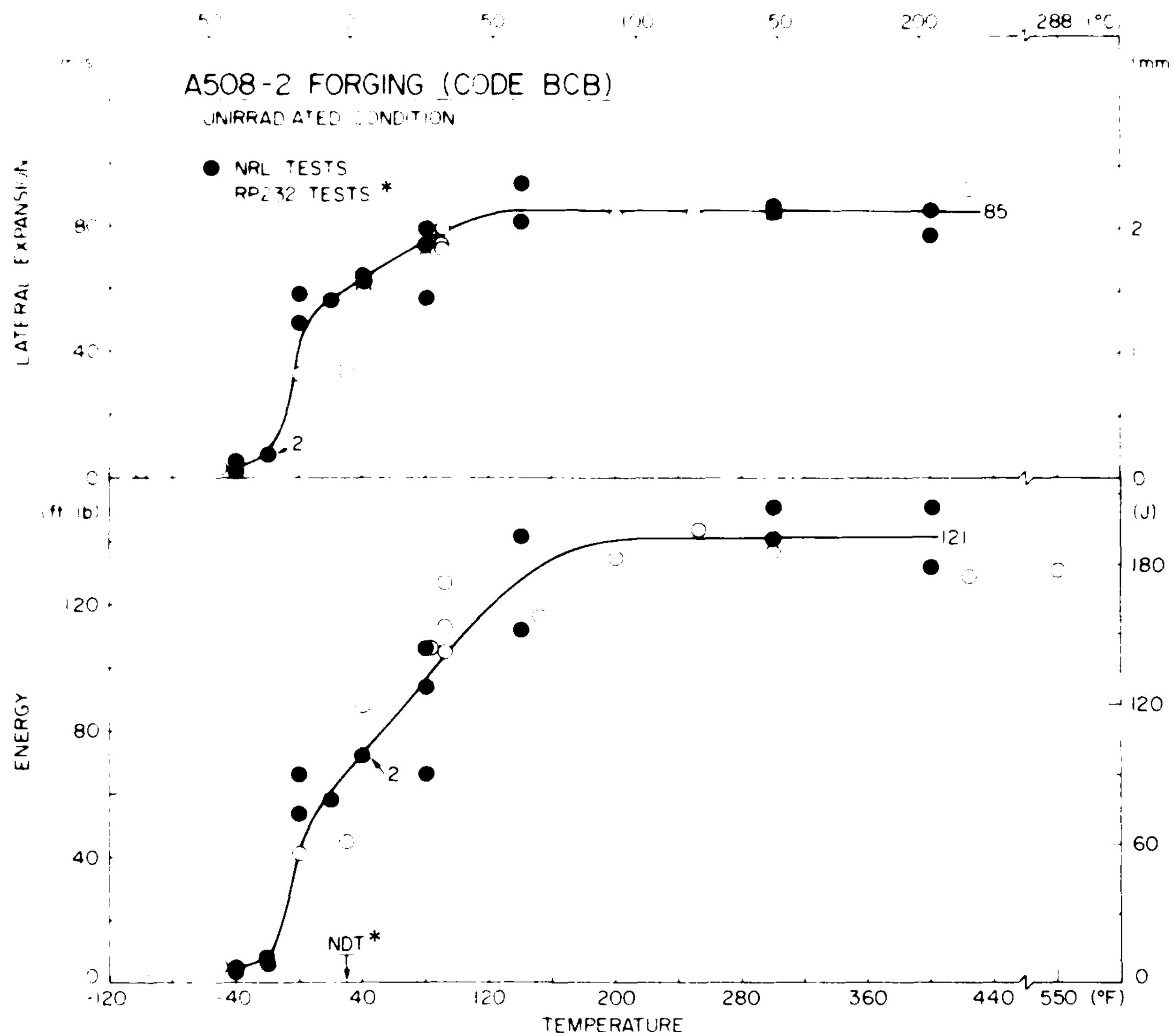

Fig. 12. Charpy-V notch ductility of the A508-2 forging, code BCB, before irradiation [2].

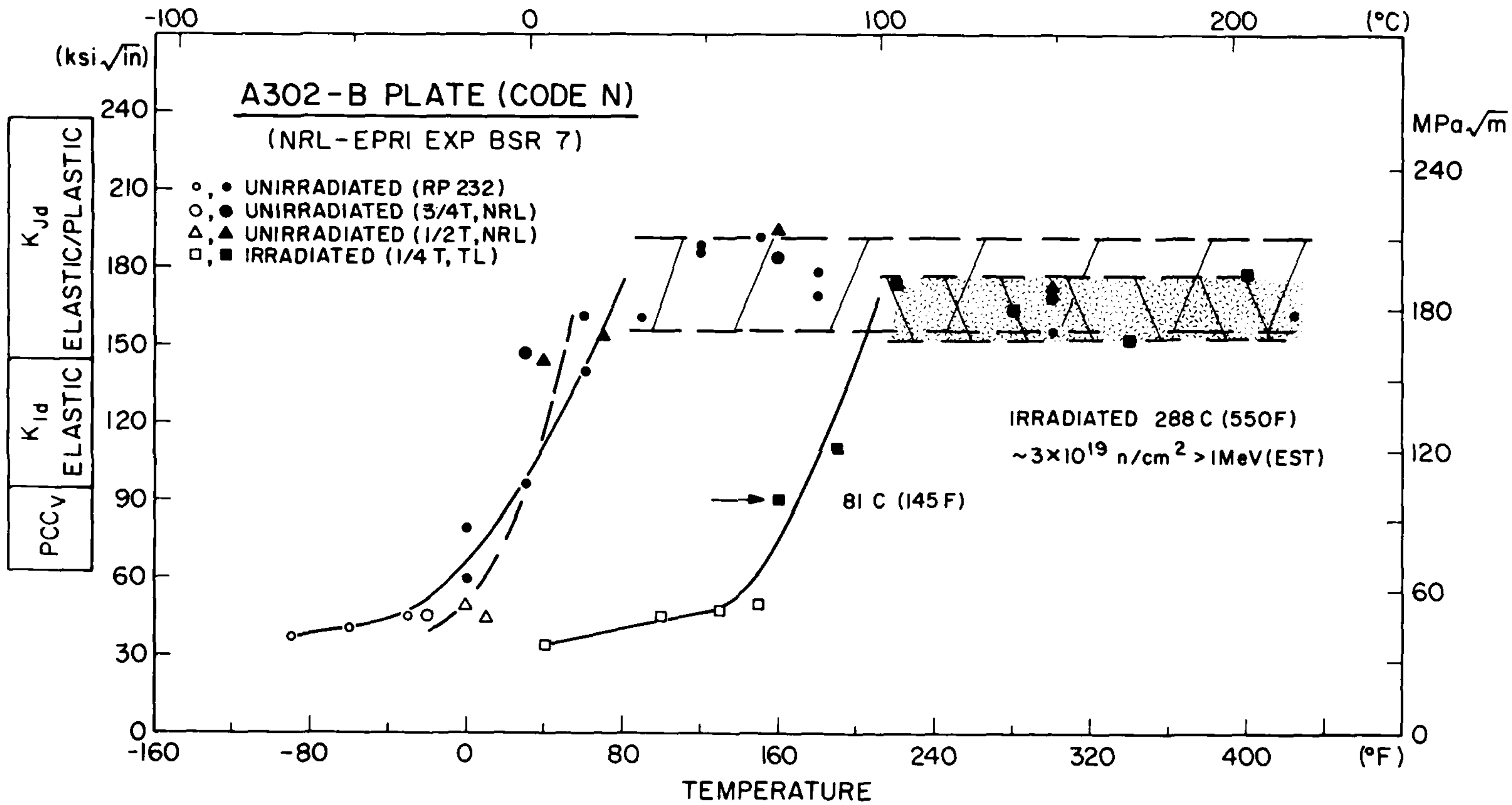

Fig. 13. Fatigue precracked Charpy-V test results for the A302-B plate, code N, before and after irradiation (experiment BSR-7).

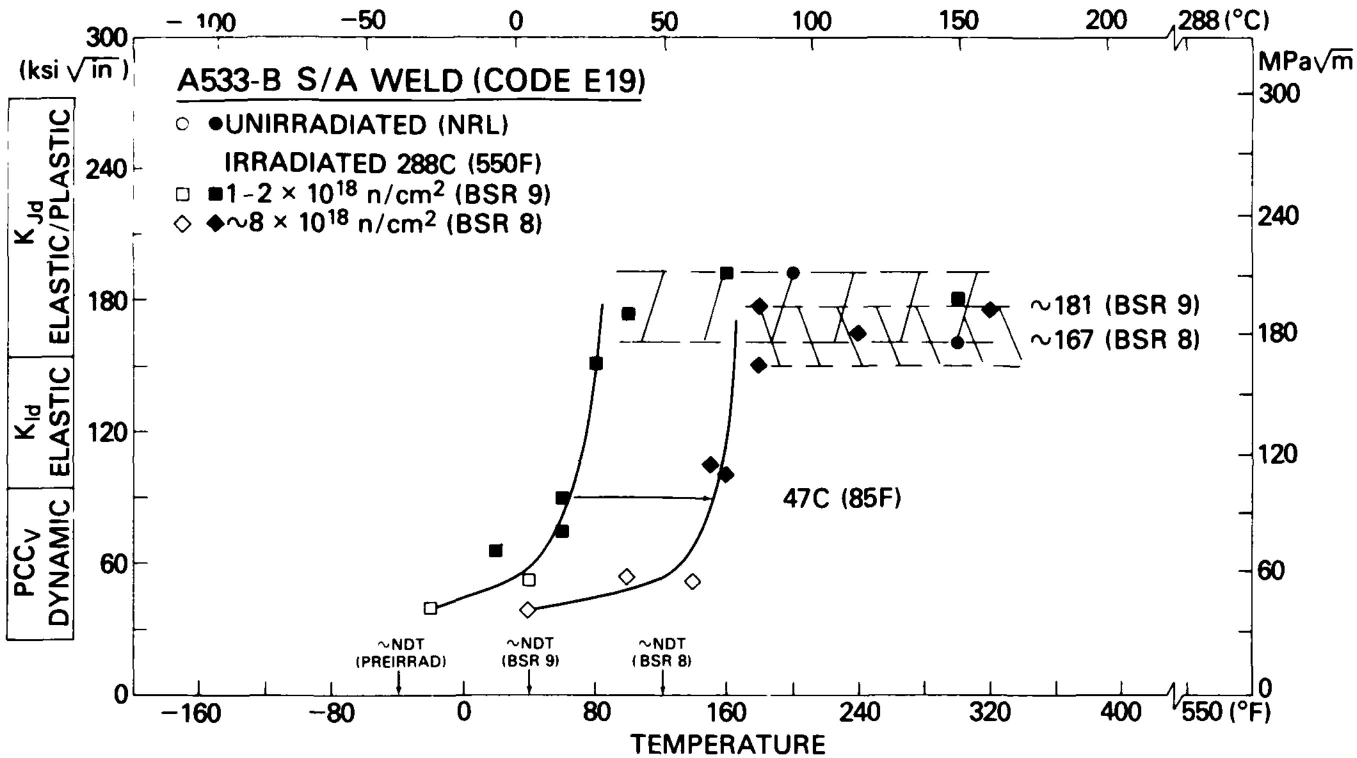

Fig. 14. Fatigue precracked Charpy-V test results for the submerged arc weld, code E19, after irradiation to two fluence levels (experiments BSR-8 and BSR-9). Two datum for the preirradiation condition are also shown.

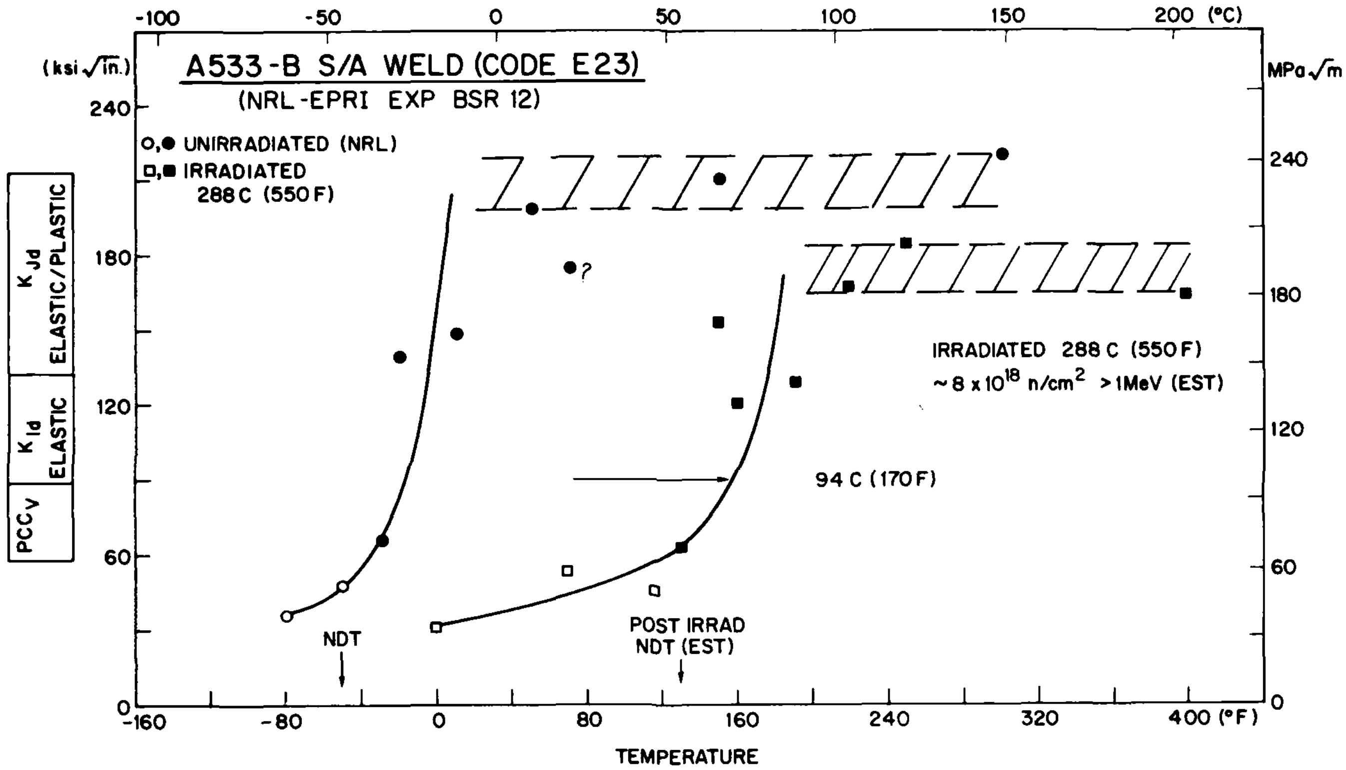

Fig. 15. Fatigue precracked Charpy-V test results for the submerged arc weld, code E23, before and after irradiation (experiment BSR-12).

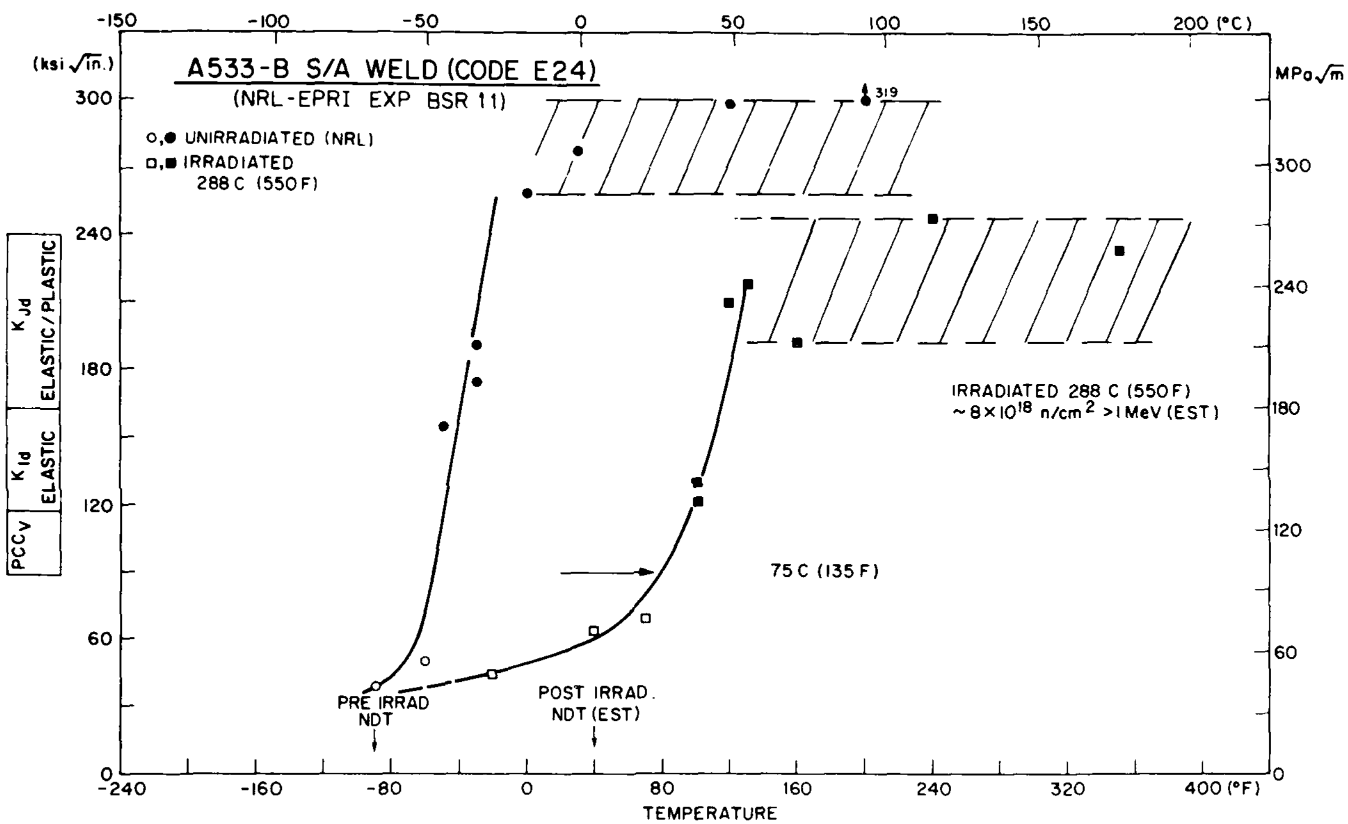

Fig. 16. Fatigue precracked Charpy-V test results for the submerged arc weld, code E24, before and after irradiation. (experiment BSR-11)

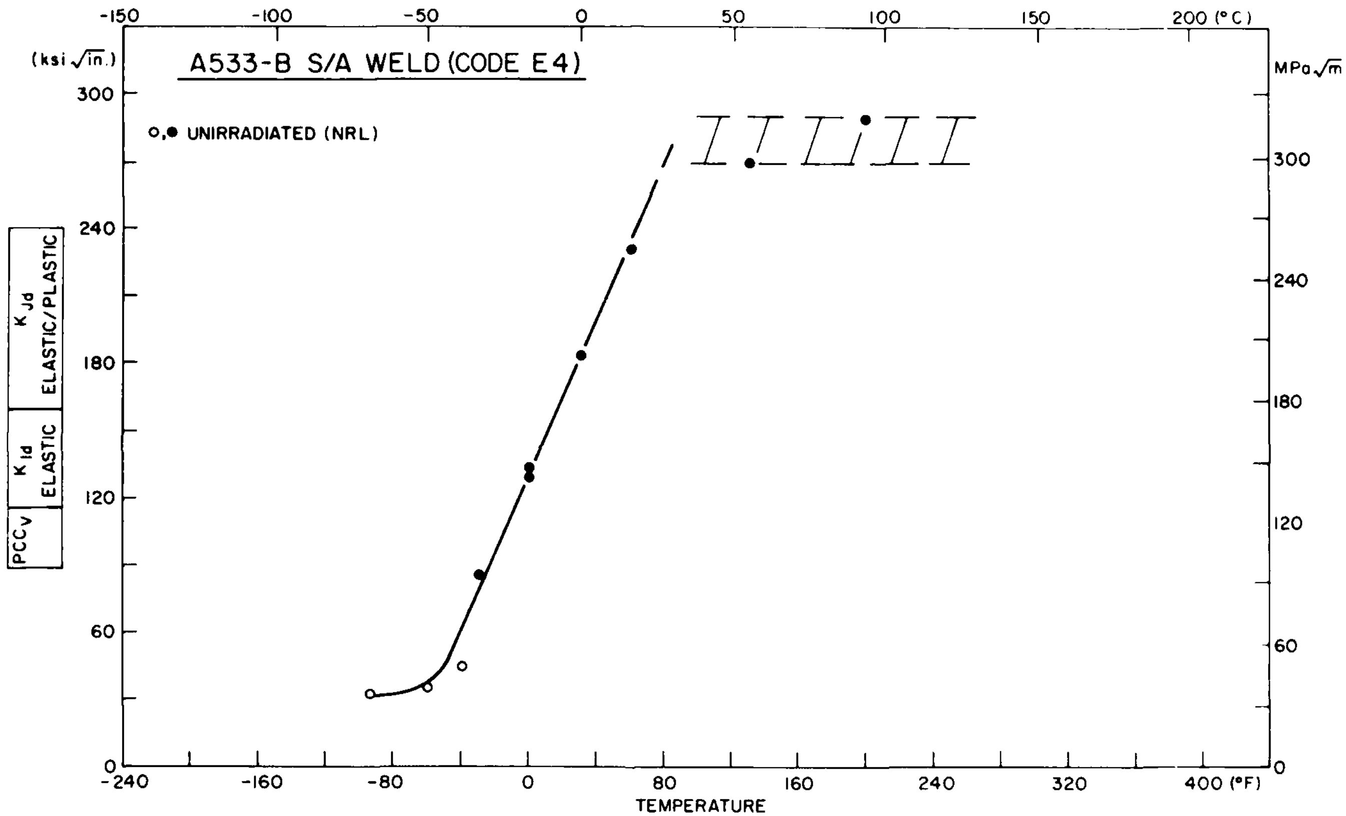

Fig. 17. Fatigue precracked Charpy-V test results for the submerged arc weld, code E4, before irradiation.

show a somewhat higher temperature transition than the RP232 Program tests. Here, the displacement is in the direction opposite that noted in the C_v data comparison [2]. In the case of plate CBB and forging BCB (below), C_v data sets for energy absorption versus temperature generally were in good agreement.

- Forging BCB (Figs. 10,11). In the transition region, wide data scatter is found. At a toughness level of 150 MPa $\sqrt{m}$ for example, the scatter band is about 39°C (70°F) wide. In turn, the data scatter precludes an exact determination of 66, 99, or 132 MPa$\sqrt{m}$ transition temperatures for indexing of the irradiation effect. The material tendency toward data scatter is not as evident from the RP232 program results (Fig. 11). In the previous report [2], a relatively high C_v data scatter coupled with a rapid upswing in C_v energy absorption within a narrow temperature range was shown for this material (Fig. 12). The PCC_v data variability may be a reflection of this behavior.

- A302-B plate, code N, and welds, codes E19, E24, E23 and E4 (Figs 13-17). Only two upper shelf tests of weld E19 could be conducted with specimens available in CY 1979. The remaining materials generally show uniform properties and low data scatter. For plate N, the NRL data compared to RP232 Program data, suggest a steeper transition curve (see dashed line). However, this indication could not be confirmed fully with the few specimens available (test stock limitation). Confirmation is not critical to the assessment of the 99 MPa$\sqrt{m}$ transition temperature elevation with irradiation as shown.

Irradiated Condition

In CY 1979, postirradiation C_v tests were completed for six reactor experiments. The experiments were *BSR-6 (forging BCB), BSR-7 (plate N), BSR-8 and BSR-9 (weld* E19), BSR-11 (weld E24) and BSR-12 (weld E23) (see Table 2). The test results are presented in Figure 18 and Figures 3 to 6, respectively. A C_v data summary is provided in Table 3. Neutron fluence values reported in the individual figures are preliminary estimates only. Neutron dosimeter wires (iron, nickel, cobalt-aluminum, silver-aluminum) included in each experiment are currently being analyzed. In this regard, each of the experiments employed the revised irradiation assembly design [2], that is, C_v and PCC_v specimens were comingled in a single capsule.

Analyses of the C_v data provided the following key observations:

- The upper shelf energy level of plate N was reduced to 47-J (35 ft-lb) from an initial level of 65-J (48 ft-lb) by the estimated 3 x 10^{19} n/cm^2 fluence. In an earlier NRL study [5], a fluence of 1.9 x 10^{19} resulted in a 56-J (41 ft-lb) upper shelf level; accordingly, a progressive decrease in upper shelf level with increasing fluence is indicated.

- A significant reduction in upper shelf level (to 76 ft-lb) was produced in weld E19 by the relatively low fluence of ~ 1 x 10^{18} n/cm^2. Further irradiation to ~8 x 10^{18} reduced the upper shelf level below 68-J (50 ft-lb). The 26-J (19 ft-lb) decrease in this case equates to a 30% reduction by irradiation.

- A fluence of ~8 x 10^{18} n/cm^2 also produced a large (22-J) reduction in upper shelf level for weld E23. An even larger reduction (38-J) in upper shelf level occurred in weld E24 with the same fluence; however, the much higher initial shelf level of this weld readily offsets the impact of this reduction.

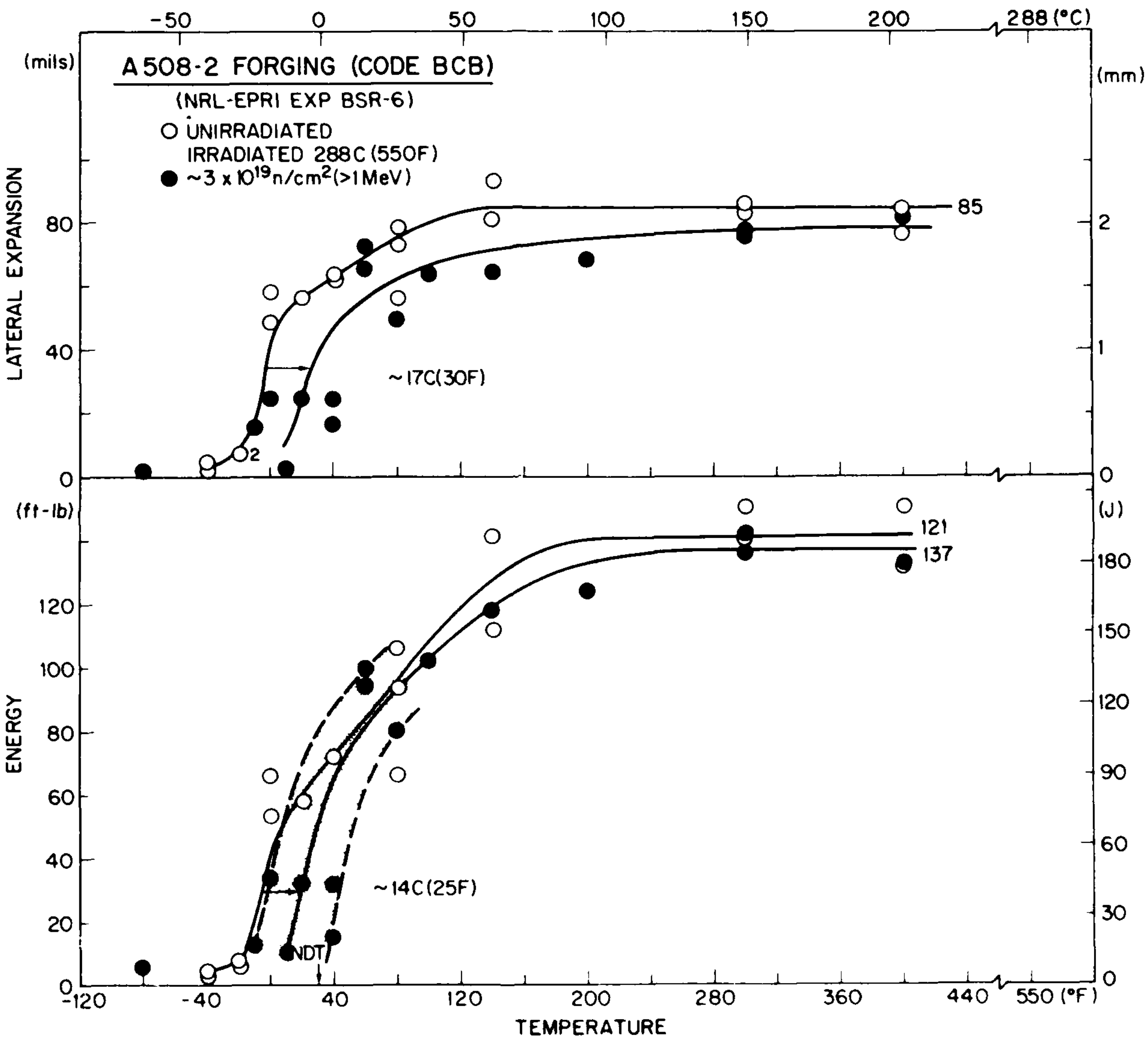

Fig. 18. Charpy-V notch ductility of the A508-2 forging, code BCB, before and after irradiation. (experiment BSR-6)

Table 3 - Preirradiation and Postirradiation Charpy-V Properties of Plate, Forging and Weld

(CY 1979 Tests)

Material/ Code	Experiment/ Fluence Φ (10^{19} n/cm^2 >1 MeV)[a]	C_v Transition Temperature						C_v Upper Shelf			
		41-J (30 ft-lb) Index		68-J (50 ft-lb) Index		0.9 mm (35 mil) Index		Energy		Lateral Expansion	
		°C	°F	°C	°F	°C	°F	J	ft-lb	mm	mils
Plate N	Unirradiated	15	60	---	---	24	75	58	43	1.07	42
	Unirradiated[b]	15	60	---	---	24	75	65	48	1.02	40
	BSR-7/≈ 3	96	205	---	---	---	---	47	35	0.81	32
	(Irrad. Change)	(81)	(145)	---	---	---	---	11	(8)	0.26	(10)
Forging BCP	Unirradiated	-21	-5	-15	5	-21	-5	191	141	2.16	85
	BSR-6/≈ 3	-7	20	-1	30	-4	25	186	137	2.01	79
	(Irrad. Change)	(14)	(25)	(14)	(25)	(17)	(30)	(5)	(4)	0.15	(6)
Weld E19	Unirradiated	-7	20	38	100	27	80	87	64	≥1.32	≥52
	BSR-9/≈ 0.1	40	105	71	160	71	160	76[c]	56[c]	1.12[c]	44[c]
	(Irrad. Change)	(47)	(85)	(33)	(60)	(44)	(80)	(11)	(8)	≥0.20	≥ 8
	BSR-8/≈ 0.8	82	180	---	---	---	---	61[c]	45[c]	0.76[d]	30[d]
	(Irrad. Change)	(89)	(160)	---	---	---	---	(26)	(19)	≥0.56	≥22
Weld E24	Unirradiated	-65	-85	-48	-55	-54	-65	179	132	2.21	87
	BSR-11/≈ 0.8	7	45	38	100	32	90	141	104	≈1.50	≈59
	(Irrad. Change)	(72)	(130)	(86)	(155)	(86)	(155)	(38)	(28)	(≈.71)	(≈28)
Weld E23	Unirradiated	-18	0	24	75	-9	15	92[c]	68[c]	1.75[c]	69[c]
	BSR-12/ ≈ 0.8	79	175	104	220	99	210	70[d]	52[d]	0.99[d]	39[d]
	(Irrad.change)	(97)	(175)	(80)	(145)	(108)	(195)	22	(16)	0.76	(30)
Weld E4	Unirradiated	-37	-35	-26	-15	-34	-30	129[d]	95[d]	2.26	89[d]
	Unirradiated	-37	-35	-26	-15	-34	-30	152[d]	112[d]	2.05	81[d]

a - Preliminary estimate
b - 1/2T Thickness location
c - 182°C (360°F)
d - 204°C (400°F)

- Transition temperature elevations (41-J index) of all three welds by ~ 8 x 10^{18} n/cm^2 are in the range of 72°C to 97°C (130°F to 175°F). Pending final dosimetry results, radiation sensitivity differences among the welds would appear to be small in terms of this property change.

- Good agreement is evident between the 41-J and 68-J transition temperature elevations. Also, the elevations in the 0.9 mm lateral expansion transition temperature are equal to or are somewhat greater than the respective 41-J transition temperature increases. Where upper shelf levels approach 68-J, differences between the three measurements will tend to be magnified by uncertainties in curve fit.

In parallel with the postirradiation C_v tests above, postirradiation PCC_v tests were completed for the same six experiments. In general, reductions in dynamic fracture toughness (K_{Jd}) at upper shelf temperatures were consisitent with the C_v upper shelf level reductions. However, the evaluation of relative effects are complicated by the PCC_v data scatter in this region in most cases. It must be recognized that, because of possible crack extension prior to development of maximum load, it is difficult to prove that values in excess of about 110 MPa$\sqrt{m}$ can still be considered as an "initiation toughness" [2].

PCC_v test results are reported in Figures 10, 11, and 13 through 16. Table 4 presents a summary of the experimental determinations. Assessments of the data trends were as follows:

- The fracture toughness transition of forging BCB (99 MPa$\sqrt{m}$ index), within the limits of the wide PCC_v data scatter, appears to have been elevated much more than the notch ductility transition described by the C_v tests (compare Figs. 18 and 10). That is, the temperature elevation by irradiation for the PCC_v data band is on the order of 53°C compared to 14°C and 17°C for the C_v curves (energy abosrption and lateral expansion). Data for 1T CT specimens of this forging, reported below, also suggest a large irradiation effect. The reason for the apparent inconsistency, which was not observed in plate or weld materials, is unclear. As pointed out above, the PCC_v and C_v specimen types were comingled in the irradiation unit and thus the inconsistency was not caused by irradiation exposure differences.

- The radiation-induced elevations in the fracture toughness transition for plate N and welds E23 and E24 agree very well with the corresponding elevations in C_v 41-J transition temperature. In the case of weld E19, reference condition data are not yet available but the difference in temperature between post-irradiation toughness transition curves (experiment BSR-8 vs BSR-9) agrees very well with the temperature difference between postirradiation C_v energy transition curves (47°C vs 42°C).

- The upper shelf dynamic fracture toughness levels of plate N, weld E19 and weld E23 after irradiation ranged from 176 to 200 MPa$\sqrt{m}$ (160 to 181 ksi$\sqrt{in}$) and correspond to C_v upper shelf values ranging from 48 to 76-J (35 to 56 ft-lb).

- Average postirradiation PCC_v upper shelf values rank the materials in the same order as average postirradiation C_v upper shelf values. However, percentage differences in dynamic fracture toughness value among materials are much less than the percentage differences in C_v energy absorption.

Table 4 - Preirradiation and Postirradiation Fatigue Precracked Charpy-V Properties of Plate, Forging and Welds

(CY 1979 Tests)

Material Code	Experiment/ Fluence ϕ (10^{19} n/cm^2 > 1 MeV)[a]	K_{Jd} Transition Temperature: 99-MPa$\sqrt{m}$ Index		66-MPa$\sqrt{m}$ Index		132-MPa$\sqrt{m}$ Index		K_{Jd} Upper Shelf	
		°C	°F	°C	°F	°C	°F	MPa$\sqrt{m}$	ksi$\sqrt{in.}$
Plate CAB	Unirradiated[b]	32	90	15	60	46	115	332-357	302-325
Plate CBB	Unirradiated[b]	10	50	-1	30	18	65	275-330	250-300
Plate N	Unirradiated[b]	-4	25	-23	-10	7	45	184-215	167-196
	BSR-7/≈ 3	77	170	66	150	88	190	167-195	152-177
	(Irrad. Change)	(81)	(145)	(89)	(160)	(81)	(145)	(~14)	(~13)
Forging BCB	Unirradiated[b]	-21/18	-5/65[c]	-32/7	-25/45[c]	-12/24	10/75[c]	324-408	295-371
	BSR-6/≈3	18/74	65/165[c]	7/63	45/145[c]	24/79	75/175[c]	345-372	314-339
	(Irrad. Change)	(~47)	(~85)	(~53)	(~95)	(~53)	(~95)	- nil -	- nil -
Weld E19[d]	BSR-9/≈ 0.1	18	65	7	45	24	75	190-211	173-192
	BSR-8/≈ 0.8	66	150	54	130	71	160	165-195	150-177
Weld E24	Unirradiated	-48	-55	-54	-65	-43	-45	282-352	257-320
	BSR-11/≈0.8	27	80	4	40	38	100	212-274	193-249
	(Irrad. Change)	(75)	(135)	(58)	(105)	(81)	(145)	(~71)	(~65)
Weld E23	Unirradiated	-26	-15	-37	-35	-23	-10	216-242	197-220
	BSR-12/≈0.8	68	155	52	125	77	170	181-203	165-185
	(Irrad. Change)	(94)	(170)	(89)	(160)	(100)	(180)	(~40)	(~36)
Weld E4	Unirradiated	-32	-25	-40	-40	-21	-5	298-319	271-290

a Preliminary estimate
b Check tests
c Approximate range
d Unirradiated condition data not available

To summarize the PCC$_v$ and C$_v$ observations made thus far, a general correlation of the two test methods is indicated by the data for plate and weld materials but not by the data for the forging material. Three of the materials show that C$_v$ upper shelf energy levels can be reduced to low levels; a 48-J postirradiation upper shelf was observed for the A302-B reference plate. Two of the welds after ~ 8×10^{18} n/cm^2 exhibited a 41-J transition temperature elevation which is greater than that for the A302-B reference plate after ~ 3×10^{19} n/cm^2; however, differences among the welds after ~ 8×10^{18} n/cm^2 are relatively small. Pending neutron dosimetry results, the C$_v$ data suggest that the radiation embrittlement sensitivity of weld E23 (intermediate copper content) is about the same as that of weld E19 (high copper content). Finally, observations made from C$_v$ and PCC$_v$ tests conducted in CY 1979 are consistent with initial observations made in CY 1978 with plates CAB and CBB [2].

Measured Versus Projected Radiation-Induced Property Change

The expression offered by the Nuclear Regulatory Commission Guide 1.99 [7] for the calculation of the C$_v$ transition temperature elevation by 288°C irradiation is:

$$T\,(^{o}F) = \left[40 + 1000\,(\%Cu - 0.08) + 5000\,(\%P - 0.08)\right](f/10^{19})^{1/2} \qquad (1)$$

where f is the fluence in n/cm^2 >1 MeV. In conjunction with this formula, the Guide imposes an upper limit on ΔT values depending on fluence level. The upper bound applies primarily to high copper content materials.

Table 5 compares elevations in C$_v$ 41-J transition temperature measured in this study against projected elevations determined by either equation (1) or the upper limit criterion as appropriate. The projections are noted to be equal or greater than the measured values in each case. Accordingly, the Guide embrittlement estimates are generally conservative. (The 6°C difference between values for plate CBB is not considered significant). Large differences between measurement and projection are apparent, on the other hand, and show a need for further refinement of current embrittlement projection capabilities. It has been proposed [6] that, along with copper and phosphorus content, nickel content be taken into account as one possible refinement since the empirical evidence suggests a reinforcement by nickel of the deterimental effect of copper on steel resistance. Plates CAB and CBB and forging BCB have nickel alloying whereas plate N and the weld deposits do not. In Table 5, the difference between measured and projected transition temperature elevation for fluences $\geq 8 \times 10^{18}$ n/cm^2 is largest for the low nickel content materials. Conversely, reference [6] shows relatively good agreement between measurement and projection for high copper, high nickel content welds for this fluence condition. The current data bank, however, may not be sufficiently comprehensive to restructure the Guide to include the nickel content variable.

THERMAL CONTROL TESTS - J. R. Hawthorne

The effects of long term 288°C heating in the absence of nuclear radiation is being evaluated for the plates CAB and CBB, the forging BCB and the production weld E4. Parallel evaluations of the plate N and the welds E19, E23 and E24 were precluded by the limited test stock. Fortunately in the case of plate N, a study of long term 302°C heating effects on C$_v$ notch ductility (TL orientation) has been made by another NRL study [8], wherein a 9726 hour exposure did not result in a significant change in either the C$_v$ 41-J transition temperature or upper shelf energy level of the material.

Table 5 - Comparison of Measured and Projected Elevations in the Charpy-V 41-J Transition Temperature by 288 °C Irradiation

Material Code	%Cu	%P	Experiment	Fluence[a] (10^{19} n/cm^2 > 1 MeV)	Elevation (°F) Experiment Measurement	Elevation (°F) Reg. Guide 1.99 Projection (Eq. (1))
			PLATE			
CAB	.12	.008	BSR-2	1.1	75	84
			BSR-3	1.5	~100	98
			BSR-5	3.0	105	139
CBB	.13	.006	BSR-4	3.0	165	156
N	.20	.015	BSR-7	3.0	145	337
			FORGING			
BCB	.036	.006	BSR-6	3.0	25	69
			WELD DEPOSIT			
E19	.385	.015[a]	BSR-9	0.1	85	105[b]
			BSR-8	0.8	160	270[b]
E24	.344	015[a]	BSR-11	0.8	130	270[b]
E23	.234	.015[a]	BSR-12	0.8	175	204

a - preliminary estimate
b - from upper limit criterion

The present thermal control test series includes all four specimen types (C_v, PCC_v, CT and tensile). However only one thermal conditioning time of 92 days (2200 hours) is being investigated. This time period corresponds approximately to the longest irradiation time employed for the four materials. The thermal conditioning treatments now stand completed. Finish machining of the C_v and tensile test specimens has been undertaken.

PLANS FOR CY 1980

Experimental plans for CY 1980 include the following efforts and objectives for program completion.

- Completion of reactor exposure of experiments BSR-10 and BSR-13 and the construction and irradiation of experiment BSR-14.

- Completion of remaining preirradiation and postirradiation C_v, PCC_v, CT and tensile tests for all materials.

- Completion of C_v and tensile tests of thermally conditioned materials. PCC_v and CT tests will be conducted only if significant notch ductility and tensile property changes are observed.

- Refinement of C_v versus PCC_v and C_v versus CT test method correlations now evolved.

- Development of final program report. The report will summarize experimental results, and test method correlations and radiation sensitivity intercomparisons permitted by the results.

ACKNOWLEDGMENTS

The authors express their appreciation to Dr. T. U. Marston of EPRI for his guidance of the research program and his assistance in securing program materials. The authors also thank L. E. Steele of NRL for his support on program administration. The important contributions of J. D. Forsyth, W. E. Hagel, T. A. Anderson and A. Hiser to certain phases of the experimental program are acknowledged also with gratitude.

REFERENCES:

1. J. R. Hawthorne, editor, "The NRL-EPRI Research Program (RP886-2), Evaluation and Prediction of Neutron Embrittlement in Reactor Pressure Vessel Materials, Annual Progress Report for CY 1977" NRL Memorandum Report 3734, Mar. 1978.

2. J. R. Hawthorne, editor, "The NRL-EPRI Research Program (RP886-2), Evaluation and Prediction of Neutron Embrittlement in Reactor Pressure Vessel Materials, Annual Progress Report for CY 1978" NRL Report 8327, August 30, 1979.

3. L. F. Miller, J. O. Johnson, F. B. K. Kam, "Characterization of the Neutron Flux in Naval Research Laboratory Irradiation Specimens at the Oak Ridge National Laboratory Bulk Shielding Reactor," letter report, Dec. 5, 1979, private communication.

4. W. L. Server and W. Oldfield, "Nuclear Pressure Vessel Steel Data Base," Fracture Control Corporation, EPRI NP 933, Dec. 1978.

5. J. R. Hawthorne, "Radiation Effects Information Generated on the ASTM Reference Correlation-Monitor Steels," ASTM DS-54, American Society for Testing and Materials, July 1974.

6. J. R. Hawthorne, "Notch Ductility Degradation of Low Alloy Steels with Low-to-Intermediate Neutron Fluence Exposures", NUREG/CR-1053, NRL Report 8357, January 14, 1980.

7. "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," Regulatory Guide 1.99, U.S. Nuclear Regulatory Commission, Office of Standards Development, Washington, D. C., Apr. 1977.

8. C. Z. Serpan, H. E. Watson, and J. R. Hawthorne, "Interaction of Neutron and Thermal Environmental Factors in the Embrittlement of Selected Structural Alloys for Advanced Reactor Applications", Nuclear Engineering and Design 11 (1970) 368-380, North-Holland Publishing Company, Amsterdam.


8